Southern Nuclear

**Cheryl A. Gayheart**
Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5316 tel
205 992 7601 fax
cagayhea@southernco.com

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Joseph M. Farley Nuclear Plant – Units 1 and 2
Final Supplemental Response to NRC Generic Letter 2004-02

Ladies and Gentlemen:

The purpose of this submittal is to provide the Southern Nuclear Operating Company (SNC) final supplemental response for Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, to Generic Letter (GL) 2004-02, dated September 13, 2004, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors."

On May 16, 2013, SNC submitted a letter of intent per SECY-12-0093, "Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" indicating FNP would pursue Closure Option 2 – Deterministic of the SECY recommendations (refinements to evaluation methods and acceptance criteria) (ML13137A131). The final outstanding issue for FNP with respect to GL 2004-02 is the in-vessel downstream effects evaluation, which addresses that long-term core cooling (LTCC) can be adequately maintained for all postulated accident scenarios that require sump recirculation.

The in-vessel downstream effects evaluation has been completed for FNP, Units 1 and 2, and is documented in the enclosure to this letter. This satisfies the GSI-191 commitment identified in the May 16, 2013 Closure Option letter.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 23rd day of March 2021.

Respectfully submitted,

Cheryl Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/RMJ

Enclosure: FNP Final Supplemental Response to GL 2004-02

cc: Regional Administrator, Region II
NRR Project Manager – Farley
Senior Resident Inspector – Farley
Director, Alabama Office of Radiation Control
RTYPE: CFA04.054

Joseph M. Farley Nuclear Plant
Final Supplemental Response to NRC Generic Letter 2004-02

Enclosure

FNP Final Supplemental Response to GL 2004-02

ENCLOSURE – FNP FINAL SUPPLEMENTAL RESPONSE TO GL 2004-04

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## 1.0 Overall Compliance

NRC Issue:

> *Provide information requested in GL 2004-02, "Requested Information." Item 2(a) regarding compliance with regulations. That is, provide confirmation that the* [Emergency Core Cooling System (ECCS)] *ECCS and* [Containment Spray System (CSS)] *CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made and this licensing basis has been updated to reflect the results of the analysis described above.*

SNC Response:

In accordance with SECY-12-0093 and as identified in SNC letter to NRC dated May 16, 2013, Joseph M. Farley Nuclear Plant (FNP) elected to pursue GSI-191 Closure Option 2 – Deterministic and identified in-vessel downstream effects as the last outstanding issue. Topical Report (TR) WCAP-17788-P, Rev. 1 provides evaluation methods and results to address in-vessel downstream effects. As discussed in NRC “Technical Evaluation Report of In-Vessel Debris Effects,” (ADAMS Accession No. ML19178A252), the NRC staff has performed a detailed review of WCAP-17788-P. Although the NRC staff did not issue a Safety Evaluation for WCAP-17788, as discussed further in “U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses” (ADAMS Accession No. ML19228A011), the staff expects that many of the methods developed in the TR can be used by PWR licensees to demonstrate adequate long-term core cooling (LTCC). Completion of the analyses demonstrate compliance with 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power plants,” (b)(5), “Long-term cooling,” as it relates to in-vessel downstream debris effects for FNP.

## 1.1 Overview of FNP Resolution to GL 2004-02

On February 28, 2008 and April 29, 2008, SNC submitted Supplemental Response to GL 2004-02 for FNP, which summarized implemented changes to support resolution of GSI-191. On December 17, 2008, SNC submitted a letter to the NRC confirming the outstanding changes identified in the April 29, 2008 letter have been implemented. The implemented changes identified in the FNP Supplemental Response remain valid and no further changes are being made to address in-vessel downstream effects.

The February 28, 2008 FNP Supplemental Response to GL 2004-02 also identified some of the conservatisms in the approach to resolving GSI-191. This list of conservatisms was further supplemented by SNC letter dated July 27, 2009. The conservatisms identified in the SNC letters remain valid, and some of the conservatisms in the in-vessel downstream effects analysis are as follows:

- The amount of fibrous debris calculated to penetrate the sump strainer is based on the NEI clean plant criteria. Compared to strainer penetration testing results, the 45%

bypass fraction used in the clean plant criteria is conservative and results in more fibrous debris penetrating the sump strainer than would be expected.

- The in-vessel fibrous debris acceptance criterion is based on the core inlet debris limit from WCAP-17788-P, Rev. 1. This assumes that all fibrous debris that enters the reactor vessel will accumulate at the core inlet. Some fraction of fibrous debris will penetrate the core inlet or bypass the core inlet via alternate flow paths.

## 1.2 Correspondence Background

The following provides a listing of correspondence issued by the NRC or submitted by SNC for FNP, on GL 2004-02:

| **Generic Letter 2004-02 Correspondences** | | |
|---|---|---|
| **Document Date** | **ADAMS Accession Number** | **Document** |
| September 13, 2004 | ML042360586 | NRC GL 2004-02 |
| February 25, 2005 | ML050610168 | First Response to GL 2004-02 |
| August 31, 2005 | ML052430746 | Second Response to GL 2004-02 |
| February 9, 2006 | ML060370387 | First NRC RAI |
| November 21, 2007 | ML073110389 | NRC Revised Content Guide |
| February 28, 2008 | ML080660657 | Supplemental Response to GL 2004-02 |
| April 29, 2008 | ML081210452 | Final Supplemental Response to GL 2004-02 |
| December 17, 2008 | ML083530543 | Completion of GL 2004-02 Commitments |
| March 9, 2009 | ML090620117 | Second NRC RAI |
| July 27, 2009 | ML092380647 | Second RAI Response |
| March 30, 2010 | ML100900004 | Supplemental Response to GL 2004-02 |
| May 5, 2010 | ML101230051 | NRC Acceptance of Completed Activities |
| May 16, 2013 | ML13137A131 | Path Forward for Resolution |

## 1.3 General Plant System Description

FNP Units 1 and 2 are Westinghouse three loop Pressurized Water Reactor (PWR) design. The Residual Heat Removal System (RHR) (low head safety injection), Centrifugal Charging Pumps (CCP) (high head safety injection) and Containment Spray System (CSS) pumps are started following a Loss of Coolant Accident (LOCA). Initially, two RHR, two CCP, and two CSS pumps take suction from the Refueling Water Storage Tank (RWST). When the RWST level reaches the low level set point, the RHR pumps are manually stopped and are realigned to take suction from the post LOCA containment sump. Once the RHR switchover to recirculation is complete, the CVCS pumps take suction from the RHR pump discharge.

When the RWST level reaches low-low level, the CSS pumps are realigned to take suction from the containment sump. There are four independent suctions (two for RHR and two for CSS) located on elevation 105’-6” in the containment, the lowest floor elevation in the containment exclusive of the reactor cavity, and they are located outside the secondary shield wall.

The FNP Nuclear Steam Supply System is a three loop PWR. The system consists of one reactor vessel (RPV), three steam generators (SGs), three reactor coolant pumps (RCPs), one pressurizer (PZR) and the Reactor Coolant System (RCS) piping. The NSSS system is located inside a bio-shield and the reactor cavity. The area inside the bio-shield is mostly open at the lowest levels, with the exception of the reactor cavity and surrounding walls in the center, and a concrete wall between the A and C loops. The concrete wall between loops A and C has a walkway against the reactor cavity wall that allows an opening between loops A and C. The outer bio-shield walls extend from the containment base elevation of 105'-6" to El. 129'-0". There are areas of the bio-shield walls that are partially open; an inner wall extends from El. 105’-6” to 116’-3”, and an outer wall extends down from El. 129’-0” to elevation 115’-3” at some locations. Above elevation 129’-0” smaller “vaults” or “coffins” surround each loop and the associated Steam Generator and Reactor Coolant Pump. These “vaults” further narrow around the Steam Generator at El. 155’-0” and extend up to El. 166’-6”. There is also a separate “vault” for the Pressurizer that begins at El. 129’-0” and extends up to El. 181’-0”.

## 1.4 General Description of Containment Sump Strainers

As stated in FNP Supplemental Response dated February 28, 2008, the FNP containment sump strainers are General Electric modular stacked disk strainers. The strainers were installed in both Unit 1 and Unit 2. Unit 1 has the only vertically stacked strainer installed on the B-Train Containment Spray pump suction. The remaining seven strainers are horizontally stacked strainers. The strainers for FNP Unit 1 and Unit 2 are mounted to the containment floor (no sump pit) located between the bio-wall and containment outside wall. This location protects the strainers from missile impacts.

For Unit 1, each strainer assembly for both the RHR strainers and CSS A-Train strainer consists of two modular horizontal stacked strainer sub-units connected to the post-

LOCA pump suction through piping. The CSS B-Train strainer assembly consists of three modular stacked disk strainer sub-units connected to a plenum that assists in directing flow to the post-LOCA pump suction inlet located within the plenum boundary. The Unit 1 RHR strainer assembly, either A-Train or B-Train, is composed of two strainer sub-units per sump, each consisting of 22 stacked disks that are 40” X 40” and provide a total of approximately 878 ft² of perforated plate surface area. The Unit 1 CSS A-Train strainer assembly consists of one strainer sub-unit with (22) 40” X 40” stacked disks and the other with (10) 40” X 40” stacked disks, providing a total of approximately 638 ft² of perforated plate surface area. The Unit 1 CSS B-Train strainer assembly is composed of three strainer sub-units, each with (13) 30” X 30” vertical stacked disks, and provides a total of approximately 389 ft² of perforated plate surface area.

Each strainer assembly for Unit 2 RHR and CSS consists of two modular horizontal stacked disk strainers connected to the sump through piping. The RHR strainer assemblies, both A-Train and B-Train, are composed of two strainers per sump, each consisting of 22 stacked disks that are 40” X 40” and provide a total of approximately 878 ft² of perforated plate surface area. The CSS A-Train strainer assembly consists of one strainer with (22) 40” X 40” stacked disks and the other with (10) 40” X 40 stacked disks, providing a total of approximately 638 ft² of perforated plate surface area. The CSS B-Train strainer assembly is composed of two strainers, one with (10) 40” X 40” stacked disks and the other with (22) 30” X 30” disks, and provides a total of approximately 433 ft² of perforated plate surface area.

The surface areas for the containment sump strainers are summarized below.

| **FNP Containment Sump Strainer Surface Area** | |
|---|---|
| **Strainer** | **Surface Area ($ft^2$)** |
| Unit 1 RHR A- and B-Trains | 878 |
| Unit 1 CSS A-Train | 638 |
| Unit 1 CSS B-Train | 389 |
| Unit 2 RHR A- and B-Trains | 878 |
| Unit 2 CSS A-Train | 638 |
| Unit 2 CSS B-Train | 433 |

## 2.0 General Description and Schedule for Corrective Actions

NRC Issue:

*Provide a general description of actions taken or planned, and dates for each. For actions planned beyond December 31, 2007, reference approved extension requests or explain how regulatory requirements will be met as per "Requested Information" Item 2(b). That is provide a general description of and implementation schedule for all corrective actions, including any plant modifications, that you identified while responding to this generic letter. Efforts to implement the identified actions should be initiated no later than the first refueling outage starting after April 1, 2006. All actions should be completed by December 31, 2007. Provide justification for not implementing the identified actions during the first refueling outage starting after April 1, 2006. If all corrective actions will not be completed by December 31, 2007, describe how the regulatory requirements discussed in the Applicable Regulatory Requirements section will be met until the corrective actions are completed.*

SNC Response:

SNC has performed analyses to determine the susceptibility of the ECCS and CSS recirculation functions for FNP to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. These analyses conform to the greatest extent practical to the NEI 04-07 methodology (Reference 1) as approved by the NRC SE dated December 6, 2004 (Reference 2). As of February 29, 2008, SNC has completed the following GL 2004-02 actions, analyses and modifications:

- NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containment"
- Latent Debris Walkdowns
- Debris Generation Analysis
- Containment Debris Transport Analysis (includes Computational Fluid Dynamics)
- Head Loss Analysis
- Hydraulic Model of the ECCS System
- CSS and RHR Net Positive Suction Head Analysis
- Vendor's Strainer Head Loss Testing
- Strainer Bypass (Penetration) Testing
- Downstream Wear and Blockage Analysis
- Chemical Effects Testing (Bench Top and Head Loss Testing)
- ECCS Throttle Valves Wear and Blockage Evaluation
- Detailed Structural Analysis of Strainers
- ECCS Sump Strainers Replacement Modification Installed
- ECCS Throttle Valves Modification Installed and Tested on Unit 1

The following are GL 2004-02 actions, analyses and modifications made by SNC following February 29, 2008:

- ECCS Throttle Valves Modification Installed and Tested on Unit 2
- Evaluation of In-vessel Downstream Effects

SNC has no outstanding corrective actions associated with GL 2004-02 for FNP.

## 3.0 Specific Information for Review Areas

As stated in FNP Supplemental Response dated February 28, 2008 and amended on April 29, 2008, and as supplemented in responses submitted on December 17, 2008, July 27, 2009, and March 30, 2010, FNP has addressed review areas 3.a through 3.m, and only the outstanding review areas 3.n through 3.p are addressed in this submittal.

## 3.n Downstream Effects – Fuel and Vessel

NRC Issue:

*The objective of the downstream effects, fuel and vessel section is to evaluate the effects that debris carried downstream of the containment sump screen and into the reactor vessel has on core cooling.*

- *Show that the in-vessel effects evaluation is consistent with, or bounded by, the industry generic guidance (WCAP-16793), as modified by NRC staff comments on that document. Briefly summarize the application of the methods. Indicate where the WCAP methods were not used or exceptions were taken, and summarize the evaluation of those areas.*

SNC Response:

Topical Report WCAP-17788-P, Rev. 1 provides evaluation methods and results to address in-vessel downstream effects. As discussed in NRC “Technical Evaluation Report of In-Vessel Debris Effects,” (ADAMS Accession No. ML19178A252), the NRC staff has performed a detailed review of WCAP-17788-P. Although the NRC staff did not issue a Safety Evaluation for WCAP-17788-P, as discussed further in “U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses” (ADAMS Accession No. ML19228A011), the staff expects that many of the methods developed in the TR may be used by PWR licensees to demonstrate adequate LTCC. SNC used methods and analytical results developed in WCAP-17788-P, Rev. 1 to address in-vessel downstream debris effects for FNP and has evaluated the applicability of the methods and analytical results from WCAP-17788-P, Rev. 1 for FNP.

3.n.1 Sump Strainer Fiber Penetration

SNC has applied the NEI clean plant criteria to determine the amount of fibrous debris penetrating the sump strainers for use in the downstream in-vessel debris analysis for FNP. The clean plant criteria, as applied to in-vessel effects, utilize a fiber penetration (bypass) fraction of 45% and a debris transport fraction of 75%.

Based on the clean plant criteria values for debris transport fraction to the strainer (T) and fiber penetration fraction (P), the following Farley specific in-vessel debris load is computed using:

$$\frac{g}{FA} = \frac{M \cdot T \cdot P}{N}$$

Where:

| | |
|---|---|
| g/FA | grams of fiber per fuel assembly |
| M | mass of fibrous debris (latent + generated from worst-case break) [grams] |
| T | transport fraction to the strainer |
| P | strainer penetration fraction |
| N | number of fuel assemblies |

The mass of latent fibrous debris for Farley is conservatively assumed to be 30 lbm. The worst-case amount of fibrous debris generated for Farley is 1 $ft^3$ of Temp-mat. The density of Temp-mat is 11.8 $lbm/ft^3$ which results in a worst-case fibrous debris generation of 11.8 lbm.

$$\frac{g}{FA} = \frac{\left(41.8\ lbm \times 453.6\frac{g}{lbm}\right) \cdot 0.75 \cdot 0.45}{157} = 40.8\ g/FA$$

This is the FNP specific in-vessel fiber load that will be compared to the applicable WCAP-17788-P, Rev. 1 in-vessel debris acceptance criterion, which assumes that all fibrous debris calculated to penetrate the strainer will reach the reactor core.

Additional discussion is required for the assumed 45% fiber penetration bypass and 75% debris transport fractions.

FNP has utilized information from the docketed responses for Vogtle Electric Generating Plant (Reference 3) to justify the fiber penetration bypass fraction for FNP. Table 3.n.1-1 provides a comparison of the critical parameters for the sump strainer bypass testing.

**Table 3.n.1-1 Critical Parameter Comparison for Sump Strainer Bypass Testing**

| Parameter | Vogtle Value | Farley Value | Evaluation |
|---|---|---|---|
| Strainer Manufacturer | GEH | GEH | The strainer design is by GEH for both Vogtle and Farley. The strainers have same overall design and same strainer hole size. The strainers for both Vogtle and Farley share the same overall form and function. |
| Strainer Perforation Size | 3/32-inch | 3/32-inch | |
| Strainer Area | Four strainer assemblies with 16 stacked disks per assembly providing approximately 677.6 $ft^2$ of total perforated plate surface area per RHR train. | Two strainer assemblies with 22 stacked disks per assembly providing approximately 878 $ft^2$ of total perforated plate surface area per RHR train. | Both Vogtle and Farley have similarly sized strainers. Both plant's strainers are stacked disks and are completely submerged during most LOCA scenarios. Vogtle's methodology computes the fiber bypass quantity based on the strainer area and not simply by a percentage of the total fiber load; therefore, differences in strainer area are accounted for. |
| Flow Rate through Single Strainer Train | 3700 gpm – design flow rate per train | 4500 gpm – maximum design flow rate per train | The Farley maximum design flow rate is used for comparison because higher flow rates will result higher approach velocities and larger bypass fractions and as such, will provide a bounding comparison. Even at the maximum design flow rate, the flow is comparable between the two because the Farley RHR strainers have more surface area for each train, which is also shown in the approach velocity comparison. Thus, the flow rate through the strainers is comparable when considering the strainer area and as a result it is reasonable to apply the Vogtle testing data to Farley. |
| Debris Type and Quantity [(1)] | Nukon and latent – 639.8 $ft^3$<br><br>Fire Barrier – 2.6 $ft^3$ | Temp-Mat – 1.0 $ft^3$<br><br>Latent (30lbm) – 12.5 $ft^3$ | Vogtle tested Nukon only due to the insulation make-up at the plant. The differences between Nukon and Temp-Mat fibrous debris is minimal such that the Vogtle test with Nukon can be applied to the Farley strainer. |
| Approach Velocity | 0.0122 ft/s | 0.0114 ft/s | The Vogtle tested range was 0.0043 ft/s – 0.0130 ft/s. Farley's maximum approach velocity is less than Vogtle's and is within the tested approach velocity. Thus, it is reasonable to apply the testing results to Farley. |

**Note:**

(1) Quantities defined at the RHR strainers for the overall worst-case break. The worst-case break is not necessarily the same for each fibrous debris type

The Vogtle bypass testing discussion in Enclosure 5 in Reference 3 shows in Figure 3.n.1-4 that the prompt fiber penetration (bypass) fraction is less than 45% with no fiber bed on the strainer and then quickly decays as the fiber bed forms. The total fibrous debris load at the FNP strainers assuming 75% transport is:

$$\left(41.8\ lbm \times 453.6\frac{g}{lbm}\right) \cdot 0.75 = 14220\ g$$

Reviewing Figure 3.n.1-4 at the Farley maximum strainer debris load of 14220 g shows that the prompt bypass decays to between 2 to 3 %. A review of Reference 3 confirms that the prompt bypass fraction makes up the majority of the fiber entering the reactor coolant system and that fiber shedding through the sump strainers is only a minor contributor. As such, the use of a constant 45% bypass fraction for Farley is conservative and justified.

As discussed above, the debris transport fraction of 75% from the NEI clean plant criteria is applied for FNP. FNP is a very low fiber plant and the insulation in containment primarily consists of reflective metal insulation. The latent fiber considered in the calculation of the in-vessel debris is conservatively high at 30 lbm. Plant walkdowns have indicated that the bounding amount of latent fiber is less than 19 lbm. In addition, the Temp-mat insulation considered in the calculations is maximized at 11.8 lbm even though the location of the insulation prevents all of the insulation from being a credible debris source from a pipe rupture. Thus, there is significant conservatism in the amount of fiber considered in determining the in-vessel debris load.

The NRC review of the NEI clean plant criteria contained in ML120730181 discusses numerous points supporting that the use of the 75% debris transport fraction is reasonable. The FNP containment is highly compartmentalized and as a result it is expected that there will be debris hold-up in numerous inactive volumes in containment. Certain areas of containment may collect debris from a pipe break but may not allow debris transport to the sump. Debris interceptors are installed at FNP, which would likely help to preclude some debris from reaching the sump strainers. Generally, while there will be some fine latent debris capture, namely due to settling and spray washout, the effectiveness of settling and spray washout mechanisms is not known and will only account for some latent debris capture/transport. It is likely that the majority of the fine latent debris will be left in place due to condensation drainage; directly sprayed surfaces will have the majority of fine debris removed but the retention of that debris is uncertain.

Thus, considering these aspects, it is reasonable to assume a 75% debris transport fraction for the in-vessel fibrous debris load.

3.n.2 Applicability to WCAP-17788 Methods and Analysis Results

FNP is a Westinghouse 3-loop PWR with an upflow barrel/baffle configuration. Per Section 3.0 of the NRC Staff Review Guidance (Reference 4), it is necessary to confirm that FNP is within the key parameters of the WCAP-17788-P, Rev. 1 methods and analysis. Each of the key parameters is discussed below.

3.n.3 Fuel Design

FNP uses Westinghouse 17x17 OFA fuel.

3.n.4 WCAP-17788 debris limit

The Proprietary total in-vessel (core inlet and heated core) fibrous debris limit contained in Section 6.5 of WCAP-17788-P Volume 1, Rev. 1 applies to FNP.

3.n.5 Methodology used to calculate the fibrous debris amounts
As described in Section 3.n.1 of this submittal, FNP assumes that all fibrous debris calculated to penetrate the strainer reaches the reactor vessel.

3.n.6 Confirm maximum combined amount of fiber that may arrive at the core inlet and heated core for hot leg break is below the WCAP-17788 fiber limit

As shown in the sump strainer fiber penetration section, the FNP maximum amount of fiber calculated to potentially reach the reactor vessel is 40.8 g/FA, which is less than the Proprietary in-vessel fibrous debris limit provided in Section 6.5 of WCAP-17788-P Volume 1, Rev. 1.

3.n.7 Confirmation that the core inlet fiber amount is less than the WCAP-17788-P, Rev. 1 threshold

FNP is a Westinghouse 3-loop PWR with an upflow barrel/baffle configuration with Westinghouse 17x17 OFA fuel. The applicable WCAP-17788-P, Rev. 1 core inlet fiber threshold is provided in Table 6-3 of WCAP-17788-P, Rev. 1. The core inlet fiber amount for FNP is calculated to be 40.8 g/FA, which is less than the applicable WCAP-17788-P, Rev. 1 core inlet fiber threshold.

3.n.8 Confirmation that the earliest sump switchover (SSO) time is 20 minutes or greater

As described in UFSAR Section 6.3.2.2.7, the earliest possible SSO time for FNP is 20 minutes.

3.n.9 Predicted chemical precipitation timing from WCAP-17788-P, Rev. 1, Volume 5 testing and the specific test group considered to be representative of the plant

Chemical precipitation timing is dependent on the plant buffer, sump pool pH, volume and temperature, and debris types and quantities. Table 3.n.9-1 summarizes the key chemical precipitation parameters and values for FNP and compares them to test group 43 from WCAP-17788-P, Rev. 1, Volume 5. Based on the comparison in Table 3.n.9-1, test group 43 is representative of FNP and the predicted chemical precipitation timing ($t_{chem}$) is 24 hours.

| **Table 3.n.9-1 Key Parameter Values for Chemical Precipitation Timing** | | |
|---|---|---|
| **Parameter** | **FNP Value** | **Test Group 43 Value** |
| Buffer | TSP | TSP |
| pH | 10.5 | 10.5 |
| Minimum Sump Volume ($ft^3$) | 39,922 | 37,655 |
| Max Sump Pool Temperature (°F) | 259 | 261 |
| CalSil (g) | 0 | 0 |
| E-glass (g) | 1 | 2 |
| Silica (g) | 0 | 0 |
| Mineral Wool (g) | 0 | 0 |
| Al Silicate (g) | 0 | 0 |
| Concrete (g) | 523 | 3,144 |
| Interam (g) | 0 | 0 |
| Al ($ft^2$) | 16,100 | 16,080 |
| Galvanized Steel ($ft^2$) | 18,210 | 14,174 |

The basis for the Test Group 43 aluminum surface area is 16,079.8 $ft^2$. However, using the FNP minimum sump volume and the Test Group 43 autoclave data, the equivalent aluminum is scaled up to 17,049.18 $ft^2$.

Using this value, the FNP quantity of aluminum is bounded. It is also noted that this quantity of aluminum includes aluminum sources which are either protected from containment spray or are located such that the containment spray cannot be transported to the active sump as described in ML100900004. Therefore, approximately 214 $ft^2$ of additional aluminum margin is available in the FNP value of 16,099.98 $ft^2$.

As discussed in Section 7.6 of WCAP-17788-P, Rev. 1, Volume 5, the amount of galvanized steel and insulation material does not impact $t_{chem}$; however, the amount of these materials should not exceed the maximum tested for each buffer type. The amount of insulation debris from Test Group 43 is bounded by the FNP values. The amount of galvanized steel at FNP is bounded by the amount tested for Test Group 33 and Test Group 34, which both use TSP as a sump buffer and consider greater than 55,000 $ft^2$ of galvanized steel (pages A-33 and A-34 of WCAP-17788-P, Rev. 1, Volume 5). Therefore, the FNP quantity of galvanized steel is acceptable.

Based on the above comparison, Test Group 43 is representative of FNP.

3.n.10 <u>Confirmation that chemical effects will not occur earlier than latest time to implement BAP mitigation measures</u>

As described in UFSAR Section 6.3.2.2.7, FNP performs injection realignment to mitigate the potential for boric acid precipitation no later than 7.5 hours, which is less than predicted chemical precipitation time of 24 hours.

3.n.11 WCAP-17788 $t_{block}$ value for the RCS design category

FNP is a Westinghouse 3-loop PWR with an upflow barrel/baffle configuration. Based on WCAP-17788-P, Rev. 1, Volume 1, Table 6-1, $t_{block}$ for FNP is 143 minutes.

3.n.12 Confirmation that chemical effects do not occur prior to $t_{block}$

The earliest time of chemical precipitation for FNP was determined to be 24 hours (see Section 3.n.9), which is greater than the applicable $t_{block}$ value of 143 minutes (see Section 3.n.11).

3.n.13 Plant rated thermal power compared to the analyzed power level for the RCS design category

FNP has a rated thermal power of 2821 $MW_t$, reflective of the Measurement Uprate Recapture (MUR) Power Uprate, which has been approved for implementation per NRC ADAMS Accession Number ML20121A283. FNP is a Westinghouse 3-loop PWR with an upflow barrel/baffle configuration and the applicable analyzed thermal power is 3658 as provided in WCAP-17788-P, Rev. 1, Volume 4, Table 6-1. The FNP rated thermal power is less than the analyzed power; therefore, this parameter is bounded by the WCAP-17788-P, Rev. 1 alternate flow path analysis.

3.n.14 Plant alternate flow path (AFP) resistance compared to the analyzed AFP resistance for the plant RCS design category

FNP is a Westinghouse upflow barrel/baffle plant. The Proprietary analyzed AFP resistance is provided in Table 6-1 of WCAP-17788-P Volume 4, Rev. 1. The Proprietary FNP specific AFP resistance is provided in Table RAI-4.2-24. The FNP specific AFP resistance is less than the analyzed value; therefore, the FNP AFP resistance is bounded by the resistance applied to the AFP analysis.

3.n.15 Consistency between the minimum ECCS flow per FA assumed in the AFP analyses and that at the plant

FNP is a Westinghouse upflow barrel/baffle plant. The AFP analysis for Westinghouse upflow plants analyzed a range of ECCS recirculation flow rates from 8 – 40 gpm/FA, as shown in Table 6-1 of WCAP-17788-P Volume 4, Rev. 1. The FNP ECCS recirculation flow rate corresponding to the worst-case GSI-191 hot leg break scenario is 8.9 gpm/FA which is within the range of ECCS recirculation flow rates considered in the AFP analysis.

3.n.16 Summary

The comparison of key parameters used in the WCAP-17788 AFP analysis to the FNP specific values is summarized in Table 3.n.16-1. Based on these comparisons FNP is bounded by the key parameters and the WCAP-17788 methods and results are applicable.

| Table 3.n.16-1 Key Parameter Values for In-Vessel Debris Effects | | | |
|---|---|---|---|
| **Parameter** | **WCAP-17788 Value** | **FNP Value** | **Evaluation** |
| Maximum Total In-Vessel Fiber Load (g/FA) | Volume 1 Section 6.5 | 40.8 | Maximum in-vessel fiber load is less than WCAP-17788 limit. |
| Maximum Core Inlet Fiber Load (g/FA) | Volume 1 Table 6-3 | 40.8 | Maximum core inlet fiber load is less than WCAP-17788 threshold. |
| Minimum Sump Switchover Time (min) | 20 | 21 | Later switchover time results in a lower decay heat at the time of debris arrival, reducing the potential for debris induced core uncovery and heatup. |
| Minimum Chemical Precipitate Time (hr) | 2.4 ($t_{block}$) | 24 ($t_{chem}$) | Potential for complete core inlet blockage due to chemical product generation would occur much later than assumed. |
| Maximum Hot Leg Switchover Time (hr) | 24 ($t_{chem}$) | 7.5 | Latest hot leg switchover occurs well before the earliest potential chemical product generation. |
| Rated Thermal Power ($MW_t$) | 3658 | 2821 | Lower rated thermal power results in lower decay heat. |
| Maximum AFP Resistance | Volume 4 Table 6-1 | Volume 4 Table RAI-4.2-24 | AFP resistance is less than the analyzed value, which increases the effectiveness of the AFP. |
| Minimum ECCS Recirculation Flow (gpm/FA) | 8 | 8.9 | Maximum debris bed resistance at the core inlet occurs at lower flow rates. |

## 3.o Chemical Effects

NRC Issue:

> *The objective of the chemical effects section is to evaluate the effect that chemical precipitates have on head loss and core cooling.*
>
> 1) Provide a summary of evaluation results that show that chemical precipitates formed in the post-LOCA containment environment, either by themselves or combined with debris, do not deposit at the sump screen to the extent that an unacceptable head loss results, or deposit downstream of the sump screen to the extent that long-term core cooling is unacceptably impeded.

SNC Response:

The FNP chemical effects analysis of the sump strainers was submitted in Supplemental Response dated February 28, 2008, and amended on April 29, 2008 as well as supplemental

responses submitted on July 27, 2009 and March 30, 2010. The FNP sump strainer chemical effects analysis is unchanged.

The FNP in-vessel chemical effects analysis is described in Sections 3.n.9 through 3.n.12.

## 3.p Licensing Basis

NRC Issue:

> *The objective of the licensing basis section is to provide information regarding any changes to the plant licensing basis due to the sump evaluation or plant modifications.*
>
> 1) Provide the information requested in GL 04-02 Requested Information Item 2(e) regarding changes to the plant licensing basis. The effective date for changes to the licensing basis should be specified. This date should correspond to that specified in the 10 CFR 50.59 evaluation for the change to the licensing basis.

SNC Response:

The FNP licensing basis has been changed in accordance with the requirements of 10 CFR 50.71(e) to incorporate the GL 2004-02 response, including the values of analyzed debris limits needed for TSTF-567.

## 4.0 References

1. NEI 04-07, Volume 1, Revision 0, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” December 2004. Available in NRC ADAMS Accession Number ML050550138.

2. NEI 04-07, Volume 2, Revision 0, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” December 2004. Available in NRC ADAMS Accession Number ML050550156.

3. NL-18-0915, “Vogtle Electric Generating Plant – Units 1 & 2 Supplemental Response to NRC Generic Letter 2004-02,” July 10, 2018, ADAMS Accession No. ML18193B162.

4. NRC Memorandum from V. G. Cusumano to J. E. Marshall, “U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses,” September 4, 2019, ADAMS Accession No. ML19228A011.